AmerenUE
Callaway Plant

PO Box 620
Fulton, MO 65251

May 8, 2007

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop P1-137
Washington, DC 20555-0001

ULNRC05412

Ladies and Gentlemen:

DOCKET NUMBER 50-483
CALLAWAY PLANT UNIT 1
UNION ELECTRIC CO.
FACILITY OPERATING LICENSE NPF-30
LICENSEE EVENT REPORT 2007-002
Manual Reactor Trip at Reduced Power due to Inadequate Feedwater Control

The enclosed licensee event report is submitted in accordance with 10CFR50.73(a)(2)(iv)(A) to report a manual reactor trip and auxiliary feedwater actuation.

This letter does not contain new commitments.

Sincerely,

Fadi M. Diya
Director Plant Operations

FMD/CSP/slk
Enclosure

IE22

Mr. Bruce S. Mallett
Regional Administrator
U.S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

Senior Resident Inspector
Callaway Resident Office
U.S. Nuclear Regulatory Commission
8201 NRC Road
Steedman, MO 65077

Mr. Jack N. Donohew (2 copies)
Licensing Project Manager, Callaway Plant
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-7D1
Washington, DC 20555-2738

Missouri Public Service Commission
Governor Office Building
200 Madison Street
PO Box 360
Jefferson City, MO 65102-0360

Records Center
Institute of Nuclear Power Operations
700 Galleria Parkway
Atlanta, GA 30339

bcc: C. D. Naslund
A. C. Heflin
C. R. Younie
D. T. Fitzgerald
G. A. Hughes
D. E. Shafer
S. L. Gallagher (100)
L. M. Belsky (NSRB)
K. A. Mills
P. M. Bell
msp@nrc.gov
ded@nrc.gov
LEREvents@inpo.org
babrook@wcnoc.com
A160.0761

Certrec Corporation
4200 South Hulen, Suite 422
Fort Worth, TX 76109

(Certrec receives ALL attachments as long as they are non-safeguards and public disclosed).

Send the following without attachments:

Ms. Diane M. Hooper
Supervisor, Licensing
WCNOC
P.O. Box 411
Burlington, KS 66839

Mr. Scott Bauer
Regulatory Affairs
Palo Verde NGS
P.O. Box 52034,
Mail Station 7636
Phoenix, AZ 85072-2034

Mr. Dennis Buschbaum
TXU Power
Comanche Peak SES
P.O. Box 1002
Glen Rose, TX 76043

Mr. Stan Ketelsen
Manager, Regulatory Services
Pacific Gas & Electric
Mail Stop 104/5/536
P.O. Box 56
Avila Beach, CA 93424

NRC FORM 366 (6-2004) | U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Manual Reactor Trip at Reduced Power Due to Inadequate Feedwater Control

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
03	09	2007	2007	- 002 -	00	05	08	2007	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE: 1

10. POWER LEVEL: 30%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
K.A. Mills, Supervising Engineer, Safety Analysis/Regional Regulatory Affairs	573-676-4317

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	SJ	CPOS	B455	YES	B	AA	CON		YES

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On March 9, 2007, while operating at 100 percent power, Callaway Plant experienced a main condenser tube leak. At 0824, high sodium levels were indicated in the main steam condenser, requiring power reduction to less than five percent power until all Action Levels for condenser sodium were exited. The on-shift Operations crew began reducing power to enter MODE 3 in accordance with procedural guidance. During the transition from the Main Feedwater Regulation Valves (MFRV) to the Main Feedwater Regulation Bypass Valves (MFRBV), problems were experienced controlling water level in the "C" steam generator (S/G). At 1043, with both the MFRV and MFRBV indicating closed and the "C" S/G level still increasing, the reactor was manually tripped. The level control problem was attributed to a failed I/P positioner on the MFRV. The controller was replaced with a like kind controller.

Digital rod position indication for control rod K10 failed during the event. The control rod was fully inserted. This failure was attributed to a bent connector pin in the digital rod position indication system. The connector pin was repaired and tested satisfactorily.

I. DESCRIPTION OF THE REPORTABLE EVENT

A. REPORTABLE EVENT CLASSIFICATION

10CFR50.73(a)(2)(iv)(A) – Any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B) of this section

(1) Reactor protection system (RPS) including: reactor scram or reactor trip.
(6) PWR auxiliary or emergency feedwater system.

B. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

MODE 1 at approximately 25-30 percent power with a rapid power reduction in progress due to a main condenser tube leak and subsequent high condenser sodium levels.

C. STATUS OF STRUCTURES, SYSTEMS OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

Valve AEFCV0570, "C" Main Feedwater Regulating Bypass Valve (MFRBV) was isolated for scheduled maintenance when the event started.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

On March 9, 2007, while operating at 100 percent power, Callaway Plant experienced a main condenser tube leak. At 0824, high sodium levels were indicated in the main steam condenser, requiring power reduction to less than five percent power until all Action Levels for condenser sodium were exited. The on-shift Operations crew began reducing power to enter MODE 3 in accordance with procedural guidance. During the transition from the Main Feedwater Regulation Valves (MFRV) to the Main Feedwater Regulation Bypass Valves (MFRBV), problems were experienced controlling water level in the "C" steam generator (S/G). At 1043, with both the MFRV and MFRBV indicating closed and the "C" S/G level still increasing, the reactor was manually tripped.

The "C" MFRBV was isolated for scheduled maintenance when the event started. Upon completion of the work, the valve was retested and returned to service at 0928 while the rapid power reduction was in progress.

The Balance of Plant (BOP) Operator reported the "C" MFRBV did not respond correctly during the transition from the MFRV to the MFRBV on the "C" steam generator. Indications were no change in feedwater flow was noted for corresponding change in valve demand in either open and closed directions. Feedwater flow changed to a value greater than expected causing steam generator level to rise outside the normal operating band. Feedwater flow to the "C" steam generator indicated about 1.3E6 lbs/hr and level in the "C" S/G continued to increase, while both the "C" MFRV and MFRBV indicated closed. The BOP Operator reported the condition and at 10:43 the Control Room Supervisor directed the Reactor Operator to trip the reactor due to loss of level control for "C" steam generator. (At this point it should be noted that the actual problem was a failed I/P positioner on MFRV

AEFCV0530).

After the manual reactor trip all rods indicated fully inserted except for rod K10. The digital rod position indication (DRPI) for control rod K10 failed during the trip. Control rod K10 was believed to be fully inserted, but was treated as indeterminate. Rod K10 was assumed as not inserted when the shutdown margin calculation was performed and the plant conservatively implemented the actions of FSAR specification 16.1.3.1. Troubleshooting showed that the problem with was with the DRPI for control rod K10, therefore, all rods were fully inserted after the reactor trip.

E. METHOD OF DISCOVERY OF EACH COMPONENT, SYSTEM FAILURE, OR PROCEDURAL ERROR

During the power reduction, "C" S/G level was increasing with both the MFRV and MFRBV indicating closed. Subsequent investigation indicated that the positioner for the "C" MFRV had failed. A field walkdown of AEFCV0530, following the plant trip, identified that the positioner was exhibiting abnormal indications. Observation of the LCD display indicated that "ERROR 12" was being displayed on the positioner. The positioner was periodically reinitializing on approximately 5 second intervals. It was also identified that AEFCV0530 was off its closed seat as identified by a yellow positioning pin being approximately one inch above the close limit switch actuating arm.

The rod K10 DRPI malfunction was discovered after the reactor trip, when a general warning annunciator was received for the digital position rod indication for control rod K10.

II. EVENT DRIVEN INFORMATION

A. SAFETY SYSTEMS THAT RESPONDED

All safety systems functioned as designed with the exception of the digital rod position indication for control rod K10 which failed during the trip due to an existing problem. Control rod K10 was believed to be fully inserted, but was being treated as indeterminate. Main feedwater isolation and motor-driven auxiliary feedwater actuations occurred as expected. Both trains of the emergency diesel generators and the offsite power supplies were available.

B. DURATION OF SAFETY SYSTEM INOPERABILITY

Duration of the indication problem for control rod K10 was from the time of the reactor trip on March 9, 2007, through completion of post maintenance testing on the digital rod position system for control rod K10 on March 11, 2007. The duration was 1 day 13 hours.

The safety function for the MFRV and MFRBV is to close on a feedwater isolation signal. The safety function occurred when the reactor was tripped on March 9, 2007. There was not a safety system inoperability associated with the MFRV or MFRBV.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

The "C" MFRV, AEFCV0530, was able to perform its nuclear safety function to close on a Feedwater Isolation Signal (FWIS) during this event because this safeguards signal bypasses the valve positioner. From a review of plant data after the reactor trip, the MFRV fully closed when the FWIS was generated.

This event was evaluated with the Callaway PRA model. The evaluation determined that the conditional core damage probability (CCDP) of the event was less than 1E-6; therefore, this event was of very low risk significance. Use of the PRA model to evaluate the event provides for a comprehensive, quantitative assessment of the potential safety consequences and implications of the event, including the consideration of alternative conditions beyond those analyzed in the FSAR.

III. CAUSE(S) OF THE EVENT AND CORRECTIVE ACTION(S)

During the downpower, at approximately 30% power, the positioner to AEFCV0530, "C" SG MFRV, failed such that the valve would not change position on demand. The programmed failure response is 'Fail as is'. This is supported by the fact that the feedwater flow could not be reduced below 1.3E6 lbs/hr. An as-found AOV Diagnostic test was performed that determined the positioner had failed. The positioner could be made to intermittently operate by lightly tapping on it. The positioner was sent to an independent laboratory for analysis. The root cause of the event was attributed to a defective or damaged feedback shaft gear tooth in the positioner. The gear tooth damage was most likely due to a manufacturing defect or a problem during the initial factory assembly of the positioner.

The positioner has been replaced with a like kind positioner. The functionality and operability of the MFRV were verified through the performance of diagnostic and surveillance testing prior to returning the unit to service. An extent of condition review has been completed with no additional deficiencies identified.

The cause of the digital rod position indication problem for control rod K10 was a bent pin on a cable connector. The bent pin was straightened. Post maintenance testing was completed satisfactorily.

IV. PREVIOUS SIMILAR EVENTS

A reactor trip due to a MFRV positioner failure is not a recurrence for Callaway.

V. ADDITIONAL INFORMATION

The system and component codes listed below are from the IEEE Standard 805-1984 and IEEE Standard 803A-1984 respectively.

AEFCV0530, Steam Generator C Main Feedwater Regulating Valve, IEEE System: SJ, Feedwater Component: FCV, Valve, Control, Flow

Positioner on AEFCV00530, System: SJ, Feedwater, Component: CPOS, positioner Manufacturer: ABB, Model: TZID-C

Digital Rod Position Indication, System: AA, Control Rod Drive Component: CON, connector